December 6, 2001

The Honorable Fred Thompson
United States Senate
Washington, DC 20510-4204

Dear Senator Thompson:

I am responding to your letter to Mr. Dennis Rathbun, Director of the NRC's Office of Congressional Affairs, in which you forwarded the concerns of Ms. Barbara Cornett about the structural integrity of control rod drive mechanism nozzles and other vessel head penetration (VHP) nozzles in U.S. commercial nuclear power plants. Ms. Cornett forwarded her concerns to you August 29, 2001, in an e-mail. In her e-mail, Ms. Cornett referred to an August 27, 2001, article by Mr. David Lochbaum, of the Union of Concerned Scientists, in which Mr. Lochbaum criticized the NRC for apparently failing to ensure the structural integrity of VHP nozzles in U.S. commercial pressurized water reactors (PWRs). Mr. Lochbaum specifically criticized the NRC for apparently ignoring the issue of stress corrosion cracking in the VHP nozzles of U.S. PWRs and for failing to require NRC-licensed utilities to replace the upper heads of their reactor pressure vessels. Mr. Lochbaum made his article available to the public by placing the article on the World-Wide-Web at http://www.ucsusa.org/index.html. In her e-mail to you, Ms. Cornett stated that she had found the issue discussed in Mr. Lochbaum's article to be a "very frightening situation" and asked whether you believed further action was necessary to resolve the issue.

Contrary to the implication in Mr. Lochbaum's articles, the NRC has been aggressively and effectively addressing this issue. In his article, Mr. Lochbaum specifically states that "the federal agency entrusted to ensure that our nuclear reactors run safely should not turn a blind eye to a serious safety problem." NRC became involved in the issue in the early 1990s when cracking was first reported at an overseas reactor in 1991. The NRC reviewed safety assessments performed by various PWR owners groups in evaluation of the VHP nozzles for their plant designs, and concluded, in a safety evaluation dated November 19, 1993, that PWR VHP nozzle and weld cracking was not an immediate safety concern. This was predicated on the fact that the observed cracking was only axially oriented which meant that it did not pose any structural concerns. However, NRC staff also recognized the need for licensees to perform inspections to assure that any cracking that did occur would continue to be axially oriented and would be identified early. On April 1, 1997, the staff issued Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," to all of its PWR-licensees requesting that they describe the programs they had or planned to put in place to monitor and manage cracking found in VHP nozzles. The inspections that identified the cracking at the Oconee Nuclear Station in South Carolina in February 2001 were being performed as part of the program implemented by the industry in response to the 1997 generic letter.

On August 3, 2001, the Commission issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," to address the generic safety implications of the pressure boundary leakage found at the Oconee Nuclear Station. In the Bulletin, the staff requested that licensees inform the NRC of their plans, if any, to perform augmented examinations of their VHP nozzles and provide their technical basis for the timing and method of inspection. The staff is currently in the process of evaluating the adequacy of the plant-specific responses to NRC Bulletin 2001-01 and the results of inspections that have been conducted as a result of the Bulletin to assess the need for additional inspections of a plant's VHP nozzles or other regulatory action that may be appropriate.

The NRC established a number of Web sites to keep the public informed of the recent cracking events and the NRC's and the U.S. nuclear industry's efforts to address the issue of stress corrosion cracking in the VHP nozzles of U.S. PWRs. However, due to the recent terrorist events of September 11, 2001, NRC security has removed these sites from the NRC's external public Web server in order to assess whether disclosure of the contents of these Web sites poses a threat to the common defense and security of the nation. We are therefore attaching an enclosure that discusses in detail both the issue of stress corrosion cracking in the VHP nozzles of U.S. PWRs and the actions the NRC has previously taken and is currently taking to ensure the continued structural integrity of these components.

I hope the information in this letter and the enclosure will clarify to Ms. Cornett what the NRC has done and what it is doing to address the issue of stress corrosion cracking in VHP nozzles of U.S. PWRs. You have my assurance that NRC will continue to closely monitor industry actions and take appropriate regulatory action to maintain the integrity of these components and ensure safe reactor operation.

Sincerely,

/RA/

William D. Travers
Executive Director
for Operations

Enclosure: As stated

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HISTORY OF ACTIONS TAKEN BY BOTH THE U.S. NUCLEAR REGULATORY COMMISSION AND THE U.S. NUCLEAR POWER INDUSTRY TO ADDRESS THE ISSUE OF STRESS CORROSION CRACKING IN VESSEL HEAD PENETRATION NOZZLES

The vessel head penetration (VHP) nozzles of commercial U.S. pressurized water reactors (PWRs) are fabricated from Inconel 600 (also known as Alloy 600), a material that has been demonstrated to be susceptible to cracking induced primarily by age, temperature, and stress corrosion. These nozzles are part of the reactor coolant pressure boundary, which is one of three principle barriers to the release of radioactive fission products. The VHP nozzles are joined to the reactor vessel head by J-groove welds that only partially penetrate through the head thickness (Refer to Figure. 1). Stress corrosion cracking of a VHP nozzle or the weld connecting the nozzle to the vessel head can lead to leakage from the pressure boundary. If undetected and uncorrected, this type of degradation could potentially propagate to failure of the nozzle and result in a small-break loss-of-coolant accident (LOCA) for the plant. While this is not a desirable consequence, all commercial nuclear power plants are designed to accommodate certain postulated failures, including a VHP nozzle failure. All plants have emergency core cooling systems that will quickly inject coolant into the reactor and maintain it in a safe condition.

The NRC implemented an action plan in 1991 to address primary water stress corrosion cracking (PWSCC) of U.S. VHP nozzles fabricated from Alloy 600. This action plan included a staff review of safety assessments conducted by the PWR owners groups (i.e., Westinghouse Owners Group, Combustion Engineering Owners Group, and Babcock & Wilcox Owners Group). After reviewing these assessments and examining pertinent overseas inspection findings, the NRC staff concluded, in a safety evaluation (SE) dated November 19, 1993, that PWR VHP nozzle and weld cracking was not an immediate safety concern. The staff based this conclusion on the following determinations: (1) if PWSCC were to occur in a VHP nozzle, any cracks would be predominately axial in orientation, (2) the cracks would result in leakage from the nozzle to the reactor vessel head prior to any failure, and (3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the reactor vessel closure head would occur. However, in the SE, the NRC staff also stated that it had concerns about the potential for circumferential cracking to occur in these nozzles. Circumferential cracks are of greater concern because they could lead to a separation of the nozzle from the reactor vessel, which could not occur if the cracks were only oriented axially along the length of the nozzles. The SE stated that there was a need for the industry to develop enhanced leakage monitoring methods for detecting boric acid leakage from these nozzles and that new information and events could require the staff to reassess its conclusions as to the safety significance of the issue.

By letter dated March 5, 1996, the Nuclear Energy Institute (NEI)* submitted a white paper entitled “Alloy 600 RPV Head Penetration Primary Stress Corrosion Cracking,” the purpose of

* NEI is an organization that represents utilities licensed to operation commercial U.S. nuclear power plants in their efforts to address crucial regulatory or technical issues that need to be resolved with the NRC. NEI has been instrumental in assisting the industry’s licensees in their efforts to resolve the issue of stress corrosion cracking in U.S. VHP nozzles.

which was to describe how the PWR licensees were managing the issue. On April 1, 1997, the staff issued Generic Letter (GL) 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” to all of its PWR-licensees requesting that they describe the programs they had or planned to put in place to monitor and manage cracking found in VHP nozzles and inform the Commission of their intentions, if any, to perform augmented volumetric or surface examinations of the VHP nozzles for their nuclear plants. Issuance of this generic letter was noticed in pages 17887-17888 of Volume 62, *Federal Register*, No. 70, dated Friday, April 11, 1997. The NRC encouraged the industry to address this issue both on a plant-specific basis and on a generic basis. In July 1997 the Westinghouse Owners Group, Combustion Engineering Owners Group and Babcock & Wilcox Owners Group submitted their generic responses to GL 97-01 on behalf of their member utilities. The generic responses ranked the potential for the VHP nozzles of their member plants to develop stress corrosion cracking. Later, in 1998, NEI revised the rankings and developed an integrated program for inspecting the VHP nozzles of U.S. PWRs. NEI forwarded this program to the NRC for review by letter dated December 11, 1998. In regard to implementation of this program, NEI stated that licensees owning U.S. PWRs should continue to perform required visual examinations of their vessel heads for leakage, and highly recommended that plants having the most susceptible VHP nozzles implement voluntary eddy current examinations of their nozzles. NEI also stated that this program would be modified, as necessary, based on the results of all examinations performed on U.S. VHP nozzles and any other pertinent information that could provide a basis for modifying the program. The NRC staff found this approach acceptable. The NRC documented this in a letter to NEI dated March 21, 1999.

On February 18, 2001, with Unit 3 of the Oconee Nuclear Station (Oconee Unit 3) shut down, Duke Energy Corporation (Duke) performed a visual examination of the outer surface of the unit's reactor pressure vessel (RPV) head for indications of leakage. This visual examination revealed the presence of leakage in the vicinity of nine of the 69 control rod drive mechanism nozzles (a type of VHP nozzle). Upon commencing with required ASME Code Section XI repair activities of the affected CRDM nozzles, Duke identified that the flaw indications (cracks) in two of the nozzles were larger than was originally thought and had circumferential orientations to them. The extent of the length of the circumferential portions of the cracks followed the weld profile contour and were nearly 165° in length. Duke later reported that a third CRDM nozzle at Oconee Unit 3 also had a circumferential crack approximately 45° in length.

Similar stress corrosion cracking and reactor coolant pressure boundary leakage have been reported at the other two reactor units of the Oconee Nuclear Station (i.e., at Unit 1 in November 2000, and Unit 2 in April 2001) and at the reactor unit of Arkansas Nuclear One, Unit 1 (i.e., ANO Unit 1 in February 2001). Most of the cracking has been in the axial direction. The circumferential cracking is significant in that it represents the first reported occurrence of such cracking in the VHP nozzles of U.S. PWRs and raises concerns about a potentially risk-significant condition that could affect some domestic PWRs.

The circumferential cracking reported at the Oconee site has prompted the NRC and industry to re-evaluate the validity of some of the previous technical assumptions. This re-evaluation is consistent with the conclusions of NRC’s SE of November 19, 1993, which stated the staff may need to reassess its conclusions as to the safety significance of the issue based on new relevant information or cracking events. The circumferential cracking reported at Oconee also reinforces the importance of examining the upper PWR RPV head area using techniques that are capable of detecting leakage from the VHP nozzles and their associated J-groove welds and heat-affected-zones. Presently, Section XI of the ASME Boiler and Pressure Vessel Code does not require licensees to remove RPV head insulation prior to inspecting their reactor vessel heads and VHP nozzles.

After the initial finding of circumferential cracking at Oconee Unit 3, the NRC held a public meeting with NEI and the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) on April 12, 2001, to discuss circumferential cracking issues for U.S. VHP nozzles. During the meeting, the industry representatives indicated that they were developing a generic safety assessment, recommendations for revisions of near-term inspections, and long-term inspection and flaw evaluation guidelines to address this issue. On May 18, 2001, the MRP submitted the MRP-44, Part 2, report to provide an interim safety assessment for PWSCC of Alloy 600 VHP nozzles and their associated J-groove welds. To address the experience at Oconee Nuclear Station, the MRP recommended that plants considered to be highly susceptible to this form of cracking and having fall 2001 outages should perform a visual inspection of the RPV top head capable of detecting small amounts of reactor coolant leakage similar to that observed at the Oconee units and ANO Unit 1.

On June 7, 2001, the NRC held a public meeting at which the MRP provided initial responses to questions on the MRP-44, Part 2, report that the NRC staff had identified and transmitted to the MRP on May 25, 2001. The NRC staff provided additional questions on various aspects of the MRP-44, Part 2, report in a letter to the MRP dated June 22, 2001. In this letter, the staff informed the MRP that the staff had two areas of concern with the industry's methodology provided in Topical Report MRP-44, Part 2. With respect to the first area of concern, the staff informed the MRP that the conclusion that nozzle leaks would be detectable on all vessel heads would require validation. This concern was addressed in the Bulletin and licensee responses to the Bulletin. With respect to the second area of concern, the staff informed the MRP that the conclusion that the appropriate crack growth rate for OD-initiated cracking of VHP nozzles was adequately represented by crack growth data for Alloy 600 steam generator tubes would also require validation. This concern continues to be investigated.

Based on the review of industry report MRP-44, Part 2, the NRC staff concluded that additional plant-specific information was necessary to assure that licensees were taking actions to effectively maintain the integrity of their VHP nozzles. Therefore, on August 3, 2001, the Commission issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," to address the generic safety implications of the pressure boundary leakage found at the Oconee Nuclear Station and ANO Unit 1 power plants. In the Bulletin, the staff discussed the technical aspects of plant designs that could impede the ability of visual examination methods to detect leakage from the VHP nozzles of commercial U.S. PWRs. The staff emphasized that the ability to detect reactor coolant leakage from the VHP nozzles could be limited if the visual examination methods for detecting the leakage were incapable of distinguishing between boric acid residue deposited as a result of VHP nozzle leaks and those previously deposited as a result from leakage from other sources.

In the Bulletin, the staff categorized the VHP nozzles for U.S. PWRs into four populations based on a plant's susceptibility ranking as given in Appendix B to the MRP-44, Part 2, report. For the population of plants considered as having low susceptibility based upon a susceptibility ranking of more than 30 EFPY (effective full power years) of operation from the Oconee Unit 3 condition, the staff stated that the likelihood of PWSCC degradation at these facilities was low, and that enhanced examinations beyond those required by Section XI of the ASME Code were not necessary at the present time.

For the population of plants considered as having a moderate susceptibility to PWSCC based upon a susceptibility ranking of more than 5 EFPY but less than 30 EFPY of operation from Oconee Unit 3, the staff stated that an effective visual examination capable of detecting and discriminating small amounts of leakage or boric acid deposits from 100 percent of the VHP nozzles would be sufficient to provide reasonable confidence that PWSCC degradation would be identified prior to posing an undue risk. The staff emphasized that this effective visual examination should not be compromised by the presence of insulation, existing deposits on the RPV head, or other factors that could interfere with the detection of leakage.

For the population of plants considered as having a high susceptibility to PWSCC based upon a susceptibility ranking of less than 5 EFPY of operation from the Oconee Unit 3 condition, the staff stated that the possibility for leaks to occur from a VHP nozzle at one of these facilities would dictate the need to use a qualified visual examination that would be capable of reliably detecting and accurately characterizing leakage from through-wall cracks in the VHP nozzles. With respect to an examination of this sort, the staff concluded that the qualified visual examination methods should be characterized by the following aspects: (1) that, as a result of a plant-specific demonstration, any VHP nozzle exhibiting through-wall cracking would be capable of providing a sufficient leakage path to the RPV head surface (based on the as-built configuration of the VHPs), and (2) that the effectiveness of the qualified visual examination should not be compromised by the presence of insulation, existing deposits on the RPV head, or other factors that could interfere with the detection of leakage. Absent the use of a qualified visual examination, the staff stated that a qualified volumetric or surface examination of 100 percent of the VHP nozzles (with a demonstrated capability to reliably detect cracking on the OD of a VHP nozzle) would be appropriate to provide evidence of the structural integrity of the VHP nozzles.

For the population of plants which had already identified the existence of PWSCC in the CRDM nozzles (for example, through the detection of boric acid deposits), the staff concluded there was a sufficient likelihood that the cracking of VHP nozzles will continue to occur as the facilities continue to operate, and that a qualified volumetric examination of 100 percent of the VHP nozzles (with a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle) would be an appropriate method of providing evidence of the structural condition of their VHP nozzles.

In the Bulletin, the staff requested that licensees inform the NRC of their plans, if any, to perform augmented examinations of their VHP nozzles and provide their technical basis for the timing and method of inspection. The staff required addressees to submit their responses to NRC Bulletin 2001-01 within 30 days of issuance of the Bulletin. Licensee are currently implementing their inspections plans and additional cracking has been detected in some of the VHP nozzles of Crystal River Unit 3, Three Mile Unit 1, North Anna Unit 1 and Unit 2, and Surry Unit 1. None of the cracks in these VHP nozzles were reported as through-wall circumferential cracks, although one of the cracks detected at Crystal River Unit 3 has been reported as a partial through-wall circumferential crack 90° in length. When cracking is found in these nozzles, the components are being repaired before the units are returned to service.

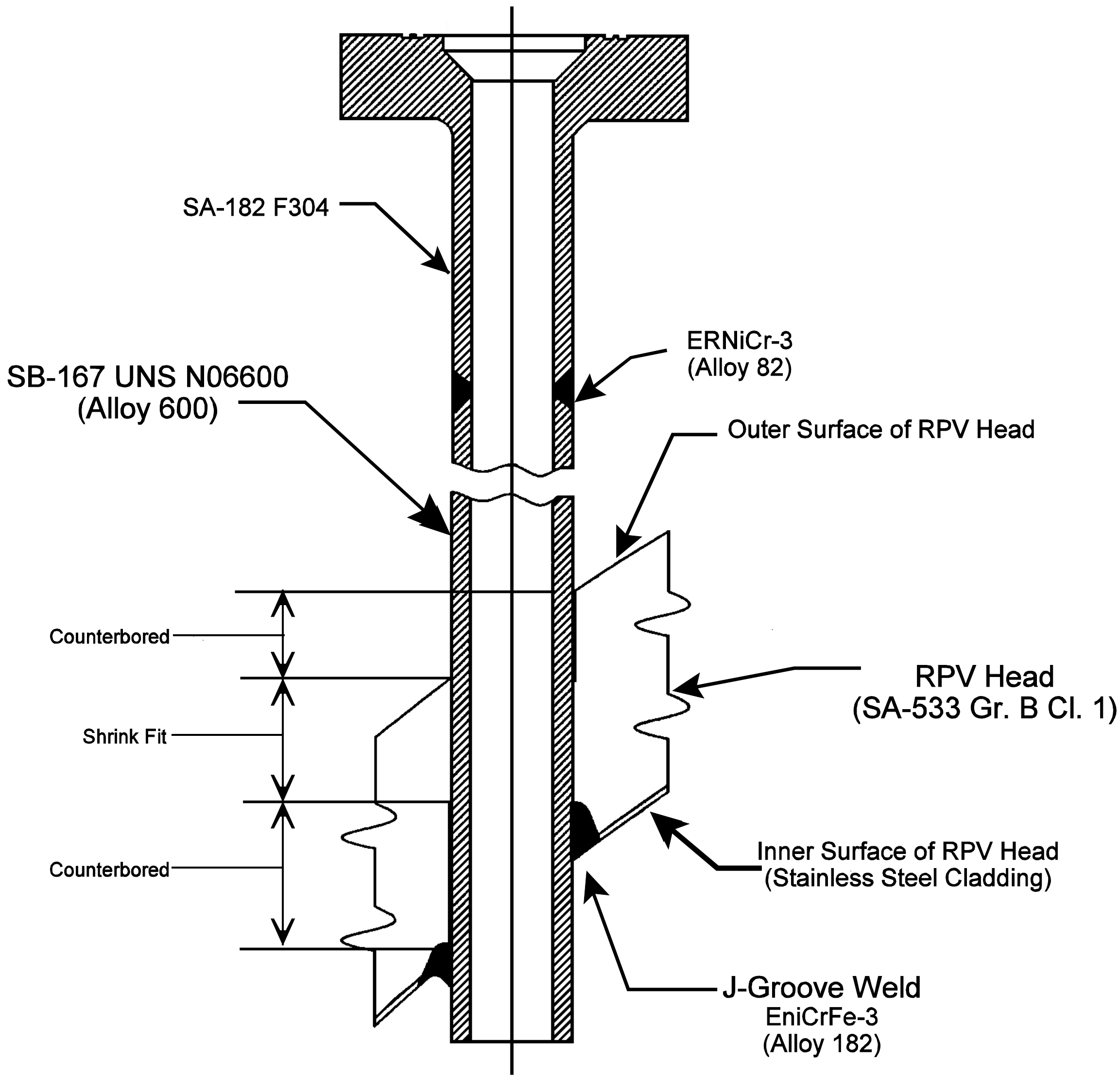

Figure 1. Schematic of Typical CRDM Nozzle Penetration